10 CFR 50.73

Boston Edison

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U.S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, D.C. 20555

Docket No. 50-293
License No. DPR-35

The enclosed Licensee Event Report (LER) 97-003-01, "Manual Scram due to Increasing Reactor Water Level During Power Reduction for Refueling Outage," is submitted in accordance with 10 CFR 50.73.

This letter contains no commitments.

Please do not hesitate to contact me if there are any questions regarding this report.

E T Boulette

E. T. Boulette, PhD

DWE/dmc/9700301

cc: Mr. Hubert J. Miller
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SUBJECT: LER 97-003-01:on 970215,manual scram occurred due to increasing RWL during power reduction for refueling outage. FW regulating valves FV-642A & FV-642B & startup FW regulating valve FV-643 were tested.W/970718 ltr.

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U.S. NUCLEAR REGULATORY COMMISSION

LICENSEE EVENT REPORT (LER)

APPROVED BY OMB NO.3150-0104
EXPIRES 4/30/98

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE(3)
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TITLE (4)
Manual Scram due to Increasing Reactor Water Level During Power Reduction for Refueling Outage

EVENT DATE (5)			LER NUMBER (6)			REPORT DATE (7)			OTHER FACILITIES INVOLVED (8)	
MONTH	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	MONTH	DAY	YEAR	FACILITY NAME N/A	DOCKET NUMBER 05000
02	15	97	97	003	01	07	18	97	FACILITY NAME N/A	DOCKET NUMBER 05000

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR: (Check one or more) (11)

OPERATING MODE (9)	N		20.402(b)		20.45(c)	X	50.73(a)(2)(iv)		73.71(b)
POWER			20.405(a)(1)(i)		50.36(c)(1)		50.73(a)(2)(v)		73.71(c)
LEVEL (10)	020		20.405(a)(1)(ii)		50.36(c)(2)		50.73(a)(2)(vii)		OTHER
			20.405(a)(1)(iii)		50.73(a)(2)(i)		50.73(a)(2)(viii)		(specify in Abstract
			20.405(a)(1)(iv)		50.73(a)(2)(ii)		50.73(a)(2)(viii)		below and in Text,
			20.405(a)(1)(v)		50.73(a)(2)(iii)		50.73(a)(2)(x)		NRC Form 366A)

LICENSEE CONTACT FOR THIS LER (12)

NAME	TELEPHONE NUMBER (Include Area Code)
Douglas W. Ellis - Principal Regulatory Affairs Engineer	508-830-8160

COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT (13)

CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS		CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS

SUPPLEMENTAL REPORT EXPECTED (14)			EXPECTED SUBMISSION DATE(15)	MONTH	DAY	YEAR
	YES (If yes, complete EXPECTED SUBMISSION DATE)	x NO				

ABSTRACT (Limit to 1400 spaces, i.e., approximately 15 single-spaced typewritten lines) (16)

On February 15, 1997, at 0038 hours, a manual scram was initiated at approximately 20 percent reactor power. The scram was intentionally initiated as a result of increasing reactor water level experienced while reducing power for the scheduled 1997 refueling outage. The scram resulted in the insertion of the control rods, transfer of the source of power to the auxiliary power distribution system, and trip of the turbine-generator.

The direct cause was greater than normal feedwater flow past the nonsafety-related feedwater system train 'A' and 'B' regulating valves at low steaming rates. Degradation of the actuator of FV-642A in conjunction with the inability of the valve's plug to travel to its lower seat was the primary root cause of the event. Corrective action taken included disassembly and repair of FV-642A and its actuator and positioner. The positioner of FV-642B was repaired and calibrated. The preventive maintenance program was modified for the train 'A' and 'B' regulating valves and startup feedwater regulating valve.

The scram was initiated when the reactor mode selector switch was moved from the RUN position to the SHUTDOWN position. The reactor vessel pressure was 954 psig with the reactor water temperature at the saturation temperature for the reactor pressure. The event posed no threat to public health and safety.

REASON FOR SUPPLEMENT

This supplement was submitted to identify the root cause of greater than normal feedwater flow past the train 'A' feedwater regulating valve FV-642A. The root cause analysis had not been completed when the original report was submitted.

BACKGROUND

In January 1997, and near the end of fuel cycle 11, the high pressure feedwater heaters were removed from heating service as planned during a power reduction on January 11 - 12, 1997. The 1997 refueling outage was scheduled to begin on February 15, 1997, and the turbine-generator was to be taken off-line at 0200 hours. Key items planned for the outage were the maintenance and testing of motor operated valves, modification of the reactor building closed cooling water system loop 'B' heat exchanger, cleanup of the suppression pool water, and the replacement of the residual heat removal and core spray systems pumps suction strainers in the suppression pool. Items pertinent to this report include the replacement of the main steam drain line valve MO-220-3.

On February 14, 1997, at 1515 hours, activities for de-inerting the primary containment atmosphere began as allowed by Technical Specification 3.7.A.1.j and 3.7.A.5. The standby gas treatment system (SGTS) train 'A' was put into service for drywell atmosphere de-inerting at 1525 hours. This action and subsequent power reduction are included in procedure 2.1.5 (rev. 51) attachment 1 section F, "Controlled Shutdown Without Manual Scram," and other procedures.

The reduction in reactor power began at 2001 hours by manually adjusting the reactor core flow via the speed controls of the recirculation system loops 'A' and 'B' pumps. The regional power authority (REMVEC) was notified of the power reduction

By 2029 hours, reactor power was 60 percent.

The feedwater system feedpump 'B' was stopped at 2042 hours. The sequential insertion of control rods began at about 2045 hours; reactor power was 56 percent, and reactor core flow was approximately 36E+06 pounds per hour at that time.

The backwash of the main condenser began at 2115 hours.

By 2145 hours, reactor core flow was further decreased by decreasing the speed of the recirculation pumps in accordance with the reactor power maneuver plan.

The operability test of the rod worth minimize (RWM) began at 2222 hours.

TEXT (If more space is required, use additional copies of NRC Form 366A) (17)

By 2315 hours, the recirculation pumps were at their minimum speed, reactor power was 30 percent, and insertion of the neutron monitoring system intermediate range monitors (IRMs) began at that time.

At 2319 hours, feedpump 'C' was stopped, and the condensate system pump 'A' was stopped at 2321 hours.

The feedwater train 'B' regulating valve (FV-642B) was closed via its controls in the control room, and the feedwater level control system was put into the single element (reactor water level) control mode at 2325 hours.

By 2329 hours, the operability test of the RWM was completed.

The SGTS train 'A' was stopped at 2332 hours, and SGTS train 'B' was started at 2335 hours for torus atmosphere de-inerting.

At 2344 hours, the insertion of the control rods was suspended to remove the low pressure feedwater heaters from service. By 2348 hours, the low pressure feedwater heaters had been removed from service and the insertion of the control rods resumed at that time. The train 'A' feedwater regulating valve FV-642A was in the automatic control mode; the train 'B' feedwater regulating valve FV-642B was closed via its controls in the control room, and the feedwater level control system was in the single element (reactor water level) control mode. The channel 'A' reactor water level transmitter LT-646A was providing the reactor water level input signal to the feedwater control system. The reactor vessel water level was at approximately +31 inches (narrow range).

An increase, at approximately two inches per minute, in reactor water level channels 'A' and 'B' was observed at 2354 hours, and a high reactor vessel water level alarm, panel C905R tile C7, occurred. Reactor water level was approximately +32 inches (increasing). Procedure 2.4.49, "Loss of Normal Feed and Feedwater Control Valve Malfunction," was entered and a limit of greater than +44 inches was chosen as the level for initiating a manual scram.

Reactor power was approximately 25 percent at the time of the observed reactor water level increase and alarm. The position of feedwater regulating valve FV-642A indicated a closed position, and feedwater flow to the reactor vessel indicated approximately 0.9 to 1.0E+06 pounds per hour. The minimum flow valves for feedpumps 'A' and 'C' were opened; feedpump 'A' was stopped, and the feedwater train 'B' block valve MO-3480 was closed. These actions were taken to reduce feedwater flow through the feedwater regulating valves (FV-642A/B) to the reactor vessel and thereby curtail the continuing, gradual increase in the reactor water level that peaked at approximately +43 inches. The reactor water level began to decrease as a result of these actions. The minimum flow valve for feedpump 'B' was not opened at that time because the valve was tagged closed.

By 0001 hours, reactor water level had decreased to approximately +30 inches. Feedpump 'C' was started at approximately 0002 hours. After the pump start, the reactor water level began to increase, at approximately two inches per minute, and continued this increase until 0008 hours.

Meanwhile, operators were directed to manually close FV-642B, located in the main condenser bay. The FV-642B position indication in the control room was closed. The manual closing of FV-642B via its local handwheel was directed because greater than normal feedwater flow through FV-642B was thought to be the cause of the increase in reactor water level. The mechanical closing of the valve via the valve's handwheel would provide additional assurance that reactor water level would not be affected by feedwater flow through the valve. Reactor water level was being maintained by FV-642A operation.

At 0008 hours, feedpump 'C' was stopped. This action was taken to eliminate the addition of feedwater to the reactor vessel because reactor water level had continued to increase. Subsequently, and with reactor water level at approximately +30 inches, a start of feedpump 'A' was initiated, but the pump motor tripped during the start sequence. Feedpump 'C' was then started. After remaining stable for a few seconds, the reactor water level began to increase, at approximately two inches per minute, and continued to increase. The senior on-shift licensed operator (NWE) was about to order a manual scram when the reactor control operator announced that reactor water level was +43 inches and decreasing. The NWE did not issue the order to initiate a scram at that time because the reactor water level was decreasing. The indicated position of valve FV-642B was closed.

By 0012 hours, the feedpump 'B' minimum flow valve had been de-tagged and opened.

Concurrently, operators had initiated actions to mechanically close valve FV-642B, and the valve was mechanically closed (gagged) by 0015 hours. The valve that supplies air to the operator of FV-642B was also closed as part of the mechanical closing of FV-642B. The rate of increase in reactor water level gradually decreased, and the level peaked, at approximately +43 inches, by 0016 hours.

By 0017 hours, reactor water level was decreasing at approximately seven inches per minute. This rate of decrease gradually slowed, and the reactor water level decrease stopped at approximately +29 inches by 0019 hours. After a brief increase of three inches, reactor water level stabilized at approximately +32 inches.

By 0030 hours, the reactor vessel water level had remained steady at approximately +32 inches for about 10 minutes. The period of steady reactor water level observed after FV-642B was mechanically closed indicated to the licensed operators that feedwater flow through FV-642B had been the cause of the reactor water level increases experienced. The insertion of the control rods resumed at that time.

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Reactor water level was again observed to be increasing, at approximately one inch per minute, at 0032 hours. This indicated to the licensed operators that reactor water level was not being effectively controlled. A pre-evolution briefing was held with the on-shift operators. The focus of the briefing was the manual initiation of a scram if reactor water level could not be controlled and the level reached +40 inches.

At 0038 hours, with the reactor water level at approximately +38 inches and continuing to increase at approximately one inch per minute, the on-shift licensed supervisor (NOS) directed the reactor control operator to initiate a manual scram.

The status of systems just prior to the event were as follows:

- Reactor power was approximately 20 percent with the reactor core flow rate at approximately 24E+06 pounds per hour. Both recirculation pumps were at the minimum speed with both controllers in the local manual control mode. The reactor vessel pressure was approximately 954 psig.

- The reactor water level was approximately +38 inches and increasing. The condensate pumps 'B' and 'C' and feedpump 'C' were in service. Feedwater flow to the reactor vessel was approximately 1E+06 pounds per hour. The feedwater train 'A' regulating valve FV-642A was in service in the automatic control mode; the train 'B' regulating valve FV-642B was mechanically closed, and the startup feedwater regulating valve FV-643 was closed. The feedwater train 'A' block valve MO-3479 was open and the train 'B' block valve MO-3480 was closed. The feedwater control system was in the single element (reactor water level) control mode with reactor water level channel 'A' transmitter LT-646A providing the reactor water level input to the feedwater control system.

- Main steam flow rate was approximately 1E+06 pounds per hour. The main turbine first stage steam pressure was approximately 89 psia (74 psig).

- The suppression pool water level was in the normal range, at approximately -4 inches (LR-5038/5049), and the bulk water temperature was approximately 70 degrees F.

The 4.16 Kv auxiliary power distribution system buses A1 through A6 were being powered from the main generator via the unit auxiliary transformer with the fast transfer control switches in the ON position. The startup transformer was in standby service. The 345 Kv transmission lines 342 and 355 were energized. The 345 Kv switchyard ring bus was energized with the switchyard air circuit breakers 102, 103, 104 and 105 in the closed position. The emergency diesel generators 'A' and 'B' were in standby service. The 23 Kv distribution system was energized. The shutdown transformer, the station blackout diesel generator, and related bus (A8) were in standby service.

EVENT DESCRIPTION

On February 15, 1997, at 0038 hours, a manually initiated reactor protection system (RPS) scram signal and scram occurred while at approximately 20 percent reactor power. The scram was the result of the intentional movement of the reactor mode selector switch from the RUN position to the SHUTDOWN position with reactor power at approximately 20 percent.

The scram signal resulted in the automatic insertion of the controls rods that had not been inserted, automatic transfer of the source of 4.16 Kv power for the auxiliary power distribution system, including emergency buses A5 and A6, from the unit auxiliary transformer to the startup transformer, and automatic trip of the main turbine-generator.

Initial control room licensed operator actions taken included the following. The feedwater train 'A' block valve MO-3479 was closed and feedpump 'C' was stopped. These actions were in accordance with procedure 2.1.6, "Reactor Scram."

Meanwhile, the reactor vessel water level decreased as expected. The decrease, to approximately +12 inches, was due to the combined effects of the decrease in the reactor water void fraction resulting from the scram and steam flow to the main condenser. The decrease, to slightly less than the low reactor water level setpoint (calibrated at approximately +12 inches), resulted in the automatic initiation of the primary containment isolation control system (PCIS) and reactor building isolation control system (RBIS) as designed.

The PCIS initiation resulted in the following responses:

- Automatic closing of the primary containment system (PCS) group 2 isolation valves that were open, including the primary containment vent and purge valves that were open for de-inerting.

- The PCS group 3/residual heat removal (RHR) system shutdown cooling suction piping isolation valves, MO-1001-47 and -50, remained closed. The RHR system low pressure coolant injection loop 'A' valve MO-1001-29A and loop 'B' valve MO-1001-29B remained closed.

- Automatic closing of the PCS group 6/reactor water cleanup (RWCU) system isolation valves MO-1201-2, MO-1201-5, and MO-1201-80.

The RBIS initiation resulted in the automatic start of the standby gas treatment system (SGTS) train 'A' and automatic closing of the secondary containment ventilation supply and exhaust dampers. SGTS train 'B', in service for torus atmosphere de-inerting, remained in operation.

Continuing control room operator response included activities for verifying the insertion of the control rods. Except for four control rods, the control rod position indicating system and video rod pattern display indicated the control rods were fully inserted. The four control rods were believed to be inserted beyond position 02. The reactor control operator confirmed the neutron monitoring system average power range monitors were downscale and reported the reactor water level was increasing. Emergency operating procedure EOP-02, "RPV Control, Failure to Scram," was entered because the insertion of the four control rods could not be verified or confirmed. EOP-01, "RPV Control," was entered earlier because the reactor water level was less than +12 inches.

After the initial decrease in the reactor water level, the reactor water level began to rapidly increase. This increase continued, and the reactor water level subsequently peaked, at approximately +65 inches, at 0056 hours. Meanwhile, the reactor vessel pressure was decreasing due to the continued flow of steam to the main condenser via the main steam piping and main turbine steam bypass system. The reactor vessel pressure decreased to approximately 800 psig by 0041 hours and then began to slowly increase.

At 0044 hours, the automatic depressurization system (ADS) was inhibited in accordance with EOP-02. The reactor water level was approximately +53 inches, and the reactor vessel pressure was approximately 840 psig at that time.

The control rod drive (CRD) system charging water valve H0-301-25 was closed at 0049 hours. This action was taken in accordance with procedure 5.3.23, "Alternate Rod Insertion."

At 0053 hours, and as the NWE was about to direct the closing of the main steam isolation valves (MSIVs), a PCIS group 1 isolation signal occurred due to high reactor water level (+54 inches) with the reactor mode selector switch not in the RUN position (SHUTDOWN) and while the reactor vessel/main steam pressure was < 810 psig (approximately 800 psig). The isolation signal resulted in the automatic closing of the MSIVs. The main steam drain isolation valves, in the closed position, remained closed. The closing of the MSIVs, with the main steam drain isolation valves already closed, eliminated the main condenser as a heat sink for reactor core decay heat generated steam and resulted in an increase in reactor vessel pressure that continued until 0101 hours. The reactor vessel water level ultimately peaked at approximately +65 inches, at 0056 hours.

Meanwhile, the insertion of all control rods was confirmed by 0056 hours. EOP-02 was exited because all control rods were confirmed to be inserted at or beyond position 02. Control rod 14-35 was at position 02.

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The residual heat removal (RHR) system loop 'A' was put into service in the suppression pool cooling mode at 0100 hours. This action was taken in anticipation of the addition of steam heat that would be introduced into the suppression pool water as a result of the intentional opening of a main steam relief valve(s) for reactor pressure control.

At 0101 hours, and at a reactor pressure of approximately 870 psig, the main steam relief valve RV-203-3B (pilot serial number 1046) was opened for approximately 40 seconds. This action was taken for reactor vessel pressure control in accordance with the guidance in EOP-01 and procedure 2.1.5. The relief valve was closed at 0102 hours, with the reactor pressure at approximately 780 psig. The opening of the relief valve resulted in a decrease in the reactor water level, to approximately +52 inches, due to the steam discharged from the reactor vessel into the suppression pool via the relief valve and its discharge piping. The reactor water level subsequently began a gradual increase.

The RHR system loop 'B' was put into service in the suppression pool cooling mode by 0102 hours.

At 0108 hours, and after the PCIS Group 6 circuitry was reset, the RWCU system was put into service in the reject mode to reduce reactor water level. In the reject mode, water from the reactor vessel is rejected to the main condenser and/or radwaste system via the RWCU system control valve CV-1239 and in-series downstream valves MO-1201-78 (to the main condenser) and/or MO-1201-77 (to the radwaste system). At 0109 hours, a high water temperature, sensed by temperature element TE-1291-13A, located on the outlet piping of the RWCU system non-regenerative heat exchanger, resulted in the automatic closing of the RWCU system isolation valve MO-1201-2 and trip of the RWCU system pump that was in service. A high water temperature sensed by TE-1291-13A, or redundant TE-1291-13B, functions to protect the RWCU system demineralizer and is not a safety-related function for containment isolation. After the Group 6 circuitry was reset, the RWCU system was returned to service at 0110 hours to reduce reactor water inventory via the reject mode.

At 0111 hours, and with the reactor pressure at approximately 880 psig, the main steam relief valve RV-203-3C (pilot serial number 1049) was opened for approximately 75 seconds. This action was taken for reactor vessel pressure control and in accordance with the guidance in EOP-01 and procedure 2.1.5. The relief valve was closed with the reactor pressure at approximately 780 psig. The opening of the relief valve resulted in a reactor water level decrease, to approximately +36 inches, due to the steam discharged from the reactor vessel into the suppression pool via the relief valve and its discharge piping. The reactor water level subsequently began a gradual increase.

At a reactor water level of approximately +42 inches, and with the reactor vessel pressure at approximately 790 psig, the high pressure coolant injection (HPCI) system was put into service in the flow test mode at 0115 hours. This action was taken for reactor pressure control and in accordance with the guidance in EOP-01. In the nonsafety-related flow test mode, steam from the reactor vessel is supplied to the HPCI turbine, and the HPCI pump circulates water from and to the condensate storage tanks. The HPCI pump is rated at 4000 gpm. The HPCI system automatically tripped at 0116 hours as a result of high reactor vessel water level (+48 inches). The reactor pressure decreased to approximately 770 psig as a result of the steam supplied to the HPCI turbine. The reactor water level increase, ultimately to approximately +55 inches, was due to the swell (expansion) of reactor vessel water resulting from the reactor vessel pressure decrease that was due to the steam supplied to the HPCI turbine. The HPCI high water level trip functions to protect the HPCI turbine and is not a safety-related function. The HPCI circuitry design includes an automatic reset of the high water level trip signal when the reactor water level is below the high water level trip setting (+48 inches).

The reactor core isolation cooling (RCIC) system was put into service in the flow test mode at 0118 hours, with the reactor water level at approximately +40 inches and the reactor pressure at approximately 800 psig. This action was taken for reactor vessel pressure control and in accordance with the guidance in EOP-01. The flow test mode of the RCIC system is nonsafety-related and similar to the flow test mode of the HPCI system. The RCIC system pump is rated at 400 gpm.

At 0125 hours, and after resetting the PCIS group 1 circuitry, the main steam drain line isolation valves MO-220-1 and MO-220-2 were opened as part of preparation activities for the opening of the MSIVs and subsequent rejection of steam heat from the reactor vessel to the main condenser. The MSIVs are pneumatically operated for the open function. Isolation valves MO-220-1, MO-220-2, and downstream drain valve MO-220-4 are part of the main steam drain piping connected upstream of the inboard MSIVs to the main condenser. Drain valve MO-220-3 is located in the drain piping connected downstream of the outboard MSIVs to the drain piping header upstream of valve MO-220-4. The opening of the drain valves is necessary for opening the MSIVs when a sufficient differential pressure exists across the MSIVs.

The outboard MSIVs were opened at 0131 hours. The in-series inboard MSIVs were not opened at that time because drain valve MO-220-3, located downstream of the outboard MSIVs and upstream of drain valve MO-220-4, could not be opened via its control switch in the control room. Operators were subsequently dispatched to manually open valve MO-220-3, located in the main condenser bay, by 0136 hours.

At 0132 hours, the ADS circuitry was reset, and the inhibit switches were returned to the normal position.

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A notification call was made to the NRC Operations Center at 0132 hours, and the NRC duty officer requested a call back in 10 minutes. The NRC Operations Center was notified of the manual scram and PCIS group 1 isolation in accordance with 10 CFR 50.72 at 0140 hours.

At 0141 hours, the RBIS was reset. The reactor water level was approximately +37 inches at that time.

The reactor water level was approximately +22 inches by 0143 hours.

The 345 Kv switchyard air circuit breakers ACB-104 and ACB-105, that had been opened at 0116 hours in accordance with REMVEC switching orders, were closed by 0143 hours. This action re-established the 345 Kv switchyard ring bus. Neither of the two sources of 345 Kv power to the startup transformer were affected while ACB-104 and ACB-105 were open.

At 0150 hours, feedpump 'B' and the startup feedwater regulating valve FV-643 were put into service.

Procedure 2.1.7, "Reactor Temperature and Pressure Checklist," was initiated at 0154 hours. The reactor vessel temperature and pressures remained within limits during the subsequent cooldown.

By 0157 hours, valve MO-220-3 had been manually opened, and the pressurizing of the main steam drain piping began.

By 0222 hours, a reactor water level band of +20 to +25 inches was established in preparation of the opening of the MSIVs.

After pressurizing the main steam drain piping and the opening of valve MO-220-4, the inboard MSIVs were opened at 0233 hours. The opening of the MSIVs was part of preparation activities for providing a steam pathway from the reactor vessel to the main condenser.

At 0236 hours, torus atmosphere de-inerting and purging resumed.

The main turbine steam sealing system was put into service at 0241 hours.

The RCIC system, in the test flow mode for reactor pressure control since 0118 hours, was returned to its normal standby lineup at 0329 hours.

At 0338 hours, the RHR system loops 'A' and 'B' were returned to the normal standby lineup.

The main turbine steam bypass valve number 1 was opened at 0345 hours. This action was taken to begin the cooldown of the reactor water through the rejection of steam heat to the main condenser via the main steam lines and main turbine steam bypass system. The three main turbine steam bypass valves are located downstream of the MSIVs and upstream of the main turbine steam stop valves. The bypass valves operate sequentially for a total steam bypass capability of approximately 25 percent of rated steam flow.

At 0401 hours, EOP-01 was exited because no condition existed that required entry into the procedure.

An automatic RPS channel 'A' trip signal occurred at 0404 hours. The trip signal was the result of a spike of the neutron monitoring system IRM 'C' (pegged high). The other five IRMs were reading less than 10 (on range 1); IRM 'C' was bypassed, and the RPS was reset.

At 0436 hours, an initial entry was made into the drywell for inspections in accessible areas of the drywell. The inspections were completed with satisfactory results by 0543.

The feedpump 'B' was removed from service at 0553 hours, and the condensate pump 'B' was removed from service at 0555 hours.

The torus atmosphere oxygen concentration was 20 percent by 0805 hours, and the SGTS train 'B' was returned to standby service.

At 0905 hours, the HPCI system automatically isolated as expected due to low reactor vessel pressure (approximately 100 psig).

The RCIC system automatically isolated as expected due to low reactor vessel pressure (approximately 75 psig) at 1006 hours.

At 1322 hours, the recirculation loop 'A' motor-generator set/pump was stopped. This action was taken in anticipation of starting the RHR system in the shutdown cooling (SDC) mode of operation. The RHR/SDC suction line is connected to the recirculation loop 'A' piping upstream of the recirculation pump suction valve. When the RHR system loop 'A' is selected for the shutdown cooling mode, shutdown cooling water flow is supplied to the reactor vessel via the recirculation loop 'A' piping downstream of the recirculation pump discharge valve. The RHR system loop 'A' was started in the shutdown cooling mode of operation with the RHR pump 'A' in service at 1350 hours.

At 1442 hours, the MSIVs were closed.

TEXT (If more space is required, use additional copies of NRC Form 366A) (17)

The reactor vessel water temperature was less than 212 degrees Fahrenheit by 1520 hours.

A post trip review was conducted in accordance with procedure 1.3.37, "Post Trip Review." A critique of the event was also conducted. The post trip review and critique included applicable personnel including the operators on shift at the time of the event.

Problem reports were written to document the scram and other observations prior to, during, or after the event and included the following: PR 97.9110 was written to document the trip of feedpump 'A' during its attempted start prior to the scram; PR 97.9111 was written to document the (perceived) problem with feedwater regulating valve FV-642B prior to the scram; PR 97.9112 was written to document the problem with the opening of valve MO-220-3 after the scram; PR 97.9113 was written to document the IRM 'C' spike after the scram; and, PR 97.0645 was written to document that a problem report was not written for the problem with the position indications of the four control rods immediately after the scram.

CAUSE

The cause of the RPS scram signal and scram was the intentional movement of the reactor mode selector switch from the RUN position to the SHUTDOWN position while the reactor power was approximately 20 percent which is greater than the high neutron flux trip setting, calibrated at less than or equal to 15 percent reactor power, for the average power range monitors (APRMs) when the mode switch is not in the RUN position. When the reactor mode selector switch was moved from the RUN position, the APRM setdown trip relays (RPS 5A-K27 series) became de-energized as designed and resulted in the expected scram signal.

The direct cause of the reactor water level increase prior to the initiation of the scram was greater than normal feedwater flow past the train 'A' feedwater regulating valve FV-642A and train 'B' feedwater regulating valve FV-642B at low steaming rates. The specific causes appear to be unique to each valve. Degradation of the actuator of FV-642A in conjunction with the inability of the valve's plug to travel to is lower seat was the primary root cause of the event. The positioner of FV-642B was degraded and out of calibration.

- The cause of the reactor water level increases prior to the scram was the inability of FV-642A to make sufficient contact with the lower seat of the valve cage. The valve was unable to make sufficient contact because of degradation of the valve actuator. After disassembly of FV-642A, it was determined the valve was not making sufficient contact with the lower seat of the valve cage. Inspection and measurement of the valve internals showed the dimension between the upper and lower cage seats was greater than the dimension between the corresponding upper and lower discs of the valve plug. Full contact was obtained on the upper seat but approximately 10 percent contact was obtained on the lower seat. The valve manufacturer indicated that, although the feedwater regulating valves are not designed to provide a tight shutoff, full contact with both seats is required to obtain a shutoff flow of less than 0.1E+06 pounds/hour. The inability of FV-642A to achieve shutoff flow was caused by degradation of the valve's actuator. The actuator should have provided sufficient force on the valve stem to cause the lower disc to make contact with the lower seat. Diagnostic testing demonstrated that the valve had a tendency to move irregularly (i.e., to stick, then jump) during the closing stroke. During the subsequent, planned overhaul (PR96.9560) of the valve actuator during the 1997 refueling outage, it was noted there was no glycol in the dampening chamber, and the lower diaphragm was degraded. The as-found condition of the actuator impaired the ability of FV-642A to modulate correctly and obtain a good shutoff condition. Valve FV-642A is a Copes-Vulcan 14" - 900 psi, double poppet, balanced, hydraulically dampened, diaphragm operated control valve equipped with a D-100-160 operator, Fisher 546 I/P controller, and Bailey AV1 positioner.

- The cause of FV-642B not reducing feedwater flow (below 0.6E+06 pounds/hour) was that the valve's positioner was degraded and out of calibration. During initial troubleshooting after the shutdown, the positioner degraded further and affected the proper venting of control air when a close signal was introduced to the valve's controls. Subsequent testing indicated the valve stroked properly and was seating properly. The valve actuator was nearly full of glycol; one pint less than the approximate 2.5 gallon capacity of the dampening chamber. It is possible that the less-than-full amount of glycol resulted in the actuator stroke to be approximately 0.1 inch short, thereby preventing the valve from fully closing. Moreover, the nearly full amount of glycol could explain why the manual closing of FV-642B resulted in a feedwater flow reduction from approximately 0.6E+06 pounds/hour to approximately 0.28E+06 pounds/hour through FV-642B. FV-642B was similar to FV-642A except for the positioner, a Bailey AP4 positioner.

The cause of the PCIS group I isolation after the scram was a high reactor water level (+54 inches) condition that occurred while the reactor mode selector switch was not in the RUN position (SHUTDOWN) with the reactor vessel/main steam pressure less than 810 psig (800 psig). The level increase was the result of decreasing reactor pressure that was due to steam flow to the main condenser via the main steam piping and main turbine steam bypass system.

CORRECTIVE ACTION

The following corrective and preventive action was taken during the 1997 refueling outage:

Feedwater regulating valves FV-642A and FV-642B, and startup feedwater regulating valve FV-643 were diagnostically tested. Air fittings were installed for future diagnostic testing.

For FV-642A:

- The valve's actuator was removed and overhauled. The overhaul included replacing the diaphragms, filling the dampening chamber with glycol, and adjusting the actuator spring.
- The valve was disassembled and repaired. The repairs included lapping and verifying full contact of the upper and lower discs and seats, adjusting the valve stroke, and repacking the valve.
- The valve's positioner was repaired and calibrated. The repair included replacement of the feedback linkage fasteners.
- The valve's air supply and related instrumentation was checked and/or calibrated.

For FV-642B:

- The valve's actuator was removed and about one pint of glycol was added to the dampening chamber. The actuator was re-installed and the actuator spring was adjusted diagnostically.
- The valve was repacked, and the valve stroke was adjusted.
- The valve's Bailey AP4 positioner was replaced with a Bailey AV1 positioner, and the new positioner was calibrated.
- The valve's air supply and related instrumentation was checked and/or calibrated.

For the startup feedwater regulating valve FV-643, the actuator spring was adjusted and the valve stroke was adjusted.

PREVENTIVE ACTION TAKEN OR PLANNED

Preventive maintenance (PM) program nodes were created and PM action items were developed to rebuild the actuators of FV-642A, FV-642B, and FV-643 every four years. The rebuild includes replacement of the actuator diaphragms, replacement of other elastomers in the actuator, and a check of the torque of actuator fasteners. The frequency and scope of the rebuild may be modified based upon experience.

Preventative Maintenance (PM) program nodes were created to perform diagnostic testing early in each refueling outage for FV-642A, FV-642B, and FV-643. The focus of the testing is to determine if internal valve maintenance or actuator maintenance is necessary. The time frame and frequency of the testing may be modified based on experience.

OTHER ACTION TAKEN

The main steam drain valve MO-220-3 was replaced during the 1997 refueling outage.

SAFETY CONSEQUENCES

This event posed no threat to the public health and safety.

The RPS scram signal was the designed response to the movement of the reactor mode selector switch from the RUN position to the SHUTDOWN position while at 20 percent reactor power.

The decrease in the reactor water level immediately after the scram was the expected response to the reactor water void fraction decrease (shrink) resulting from the scram and steam flow to the main condenser. The resultant PCIS and RBIS actuations immediately after the scram were the expected designed responses to a low reactor vessel water level condition (approximately + 12 inches).

The PCIS Group 1 isolation due to high reactor vessel water level after the scram was the designed response to a high reactor vessel water level condition.

The technical specification table 3.2.B trip setting for automatic actuation of the core standby cooling systems (CSCS) is approximately -46.3 inches. During the event, the lowest reactor vessel water level that occurred, +12 inches, was approximately 58 inches above the CSCS setpoint. In addition, the level was approximately 139 inches above the level (-127 inches) that corresponds to the top of the active fuel zone.

The CSCS consists of the HPCI system, automatic depressurization system (ADS), core spray system, and RHR system/LPCI mode. Although not part of the CSCS, the reactor core isolation cooling (RCIC) system is capable of providing water to the reactor vessel for high pressure core cooling, similar to the HPCI system. The ADS is a backup to the HPCI system and functions to reduce reactor vessel pressure to enable low pressure core cooling provided independently by the core spray system and RHR system/LPCI mode. The CSCS and the RCIC system were operable. The trip of the HPCI system was the designed response to a high reactor vessel water level condition.

The highest reactor vessel pressure that occurred was 954 psig and occurred at the time of the scram. The pressure was less than the technical specification 3.6.D setting of 1115 +/- 11 psig for the main steam relief valves and was less than the setting of 1240 +/- 13 psig for the main steam safety valves.

The pressure was less than the technical specification table 3.1.1 setting of less than approximately 1063 psig for the high reactor pressure scram function. The pressure was also less than the setpoint, calibrated at approximately 1175 psig, that initiates the ATWS system RPT and ARI functions and the setpoint, calibrated at approximately 1400 psig, that initiates the anticipated transient without scram (ATWS) system function for a feedpump trip.

The lowest reactor water level that occurred, approximately +12 inches, was greater than the setpoint, calibrated at approximately -46.3 inches, that initiates the ATWS system functions for a recirculation pump trip and alternate rod insertion.

The highest reactor vessel water level that occurred was approximately +65 inches. The level was less than the level, approximately +112 inches, corresponding to the bottom of the main steam piping.

The suppression pool bulk water temperature was not appreciably affected during the event and remained at approximately 70 degrees Fahrenheit due to the operation of the RHR system in the suppression pool cooling (SPC) mode. The temperature was less than the maximum water temperature of 120 degrees F specified by technical specification 3.7.A.1.h during reactor vessel isolation conditions.

Technical specification 3.7.A.1.m specifies the suppression pool/chamber be maintained between -6 inches and -1 inches which corresponds to a downcomer submergence of 3 feet zero inches to 3 feet five inches, respectively. The suppression pool water level was not appreciably affected during the event. The level remained at approximately -4 inches (LR-5038/5049), equivalent to approximately +127.5 inches (LI-1001-604A/B). The level was less than the level corresponding to the maximum suppression pool volume of 94,000 cubic feet specified by technical specification 3.7.A.1.b. A suppression pool volume of 94,000 cubic feet corresponds to a downcomer submergence of 4 feet, and the minimum volume of 84,000 cubic feet results in a submergence of approximately 12 inches less.

The level was less than the settings of level switches LS-2351A/B that control the automatic positioning of the suppression pool/HPCI pump suction valves.

This report is submitted in accordance with 10 CFR 50.73(a)(2)(iv) because the manual initiation of the RPS, although intentional, was not planned.

This report is also submitted in accordance with 10 CFR 50.73(a)(2)(iv) because the PCIS group 1 isolation, although a designed response to a high reactor vessel water level condition, was not planned.

SIMILARITY TO PREVIOUS EVENTS

A review was conducted of Pilgrim Station LERs submitted since 1984. The review focused on LERs submitted in accordance with 10 CFR 50.73(a)(2)(iv) that involved a reactor water level control related scram or involving a problem with a feedwater regulating valve. The review identified manual scrams reported in LERs 89-023-00, 90-013-00, and 95-003-00. Of those reports, LER 90-013-00 involved a manually initiated scram due to a blown fuse in the feedwater level control system that affected feedwater regulating valves FV-642A and FV-642B

The review also identified reactor vessel water level related automatic scrams that were reported in LERs 85-014-00, and 86-001-00, and a PCIS Group 1 isolation that was reported in LER 88-024-00. LERs 85-014-00 and 86-001-00 involved the manual control of reactor vessel water level.
LER 88-024-00 involved a high reactor water level while shut down and was due to a problem with feedwater regulating valve FV-642A. The review also identified LER 96-005-00 that involved a problem with the opening of valve MO-220-3 after a scram.

For LER 85-014-00, an automatic scram occurred on June 15, 1985, while at approximately 10 percent reactor power. The scram was the result of the automatic closing of the MSIVs, with the reactor pressure at greater than 600 psig, and was due to high reactor water level. At the time of the event, the main turbine was not in service and had been removed from service for maintenance, the reactor pressure was approximately 700 psig (decreasing); the reactor water level was being manually controlled; a high reactor water level alarm condition (+32 inches) was in alarm status and had been acknowledged; and, the reactor mode selector switch was in the STARTUP position. The cause was utility licensed operator error. The error occurred when a main turbine bypass valve was opened. The opening of the bypass valve with the reactor vessel pressure at approximately 700 psig and with water level greater than 32 inches, resulted in an increase (swell) in the reactor vessel water level. Corrective action taken included counseling operations personnel regarding the event.

TEXT (If more space is required, use additional copies of NRC Form 366A) (17)

For LER 86-001-00, an automatic scram occurred on January 6, 1986, during a plant startup at approximately 10 percent reactor power. The scram was the result of a low reactor water level condition. At the time of the event, the reactor water level was decreasing and was being manually controlled via the startup feedwater regulating valve FV-643; the main turbine was not in service; the reactor vessel pressure was approximately 930 psig; and, the reactor mode selector switch was in the RUN position. The cause was utility licensed operator error. Corrective action taken included counseling operations personnel regarding the event.

For LER 88-024-00, a PCIS Group 1 isolation occurred due to a high reactor water level while shut down on October 17, 1988, and after a flush of the feedwater system piping. The high reactor water level was caused by a pin that became disassociated from the feedback cam linkage of the valve positioner for valve FV-642A. The disconnected linkage resulted in no feedback to the valve positioner and a closed valve position indication while the valve was in an open position. Corrective action taken included the reconnection of the linkage for FV-642A.

For LER 90-013-00, a manually initiated scram occurred on September 2, 1990, while at 60 percent reactor power. The scram was initiated due to difficulties experienced in controlling reactor water level. Specifically, a fuse blew in a feedwater control circuit power supply and caused both feedwater regulating valves to lockup with no control room indication of the lockup. Corrective action taken included a modification that improved the reliability of the power supply and provides control room indication of a feedwater regulating valve lockup due to a loss of control power.

For LER 96-005-00, an automatic scram occurred on April 19, 1996, while at 22 percent reactor power. The scram was initiated by vibration in the low pressure portion of the main turbine-generator. During the subsequent post scram recovery, the main steam drain valve MO-220-3 would not open via its control switch in the control room, and the valve was subsequently opened manually. PR 96.9194 was written to document the problem with the opening of valve MO-220-3. The cause was improper shimming of the valve's limit switch gear box during the valve's replacement during the 1995 refueling outage (RFO-10). The improper shimming prevented the limit switch pinion from properly engaging with the drive sleeve bevel gear. The improper gear alignment resulted in an increased load on the drive sleeve. Corrective action taken included the replacement of the valve's limit switch cartridge and associated pinion, and inspection of the limit switch. The inspection verified no internal damage had occurred to the limit switch. The drive sleeve bevel gear was evaluated and was acceptable for use until its replacement in a subsequent outage. The valve's actuator was replaced, and the stem nut assembly was inspected with satisfactory results during the August 1996 outage. The replacement of the valve and valve actuator was scheduled for the 1997 refueling outage.

FACILITY NAME (1)	DOCKET NUMBER (2)	LER NUMBER (6)			PAGE (3)
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ENERGY INDUSTRY IDENTIFICATION SYSTEM (EIIS) CODES

The EIIS codes for this report are as follows:

COMPONENTS	CODES
Monitor (IRM)	MON
Rod (control rods)	ROD
Valve, control, flow (FV-642A/B)	FCV
Valve, electrically operated (MO-220-3)	20
Valve, relief (RV-203-3B/C)	RV

SYSTEMS	
Condensate system	SD
Containment isolation control system (PCIS)	JM
Control rod drive system	AA
Engineered safety features actuation system (RPS, PCIS)	JE
Feedwater level control system	JB
Feedwater system	SJ
High pressure coolant injection (HPCI) system	BJ
Incore monitoring system (neutron monitoring system)	IG
Main steam system	SB
Main turbine system	TA
Plant protection system (RPS)	JC
Reactor core isolation cooling system	BN